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**SODIUM GRAPHITE REACTOR
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SUMMARY

Significant items of progress in the Sodium Graphite Reactors program for this period are outlined as follows:

1. The SRE wet critical mass was determined to be 32.64 ± 0.10 fuel elements at 400° F by achieving criticality. The worth of the safety rods was determined by drop experiments. Reactivity change measurements were made on Th-U fuel elements. (Section II).
2. Studies were made of a steady-state and variable load SGR heat transfer system. This study included steam plant costs at different reactor temperatures, 850° F and 950° F steam. Results are presented. (Section I).
3. Detailed theoretical calculations on the long term reactivity changes expected in the SRE were initiated. At this time only approximate calculations of the reactivity changes have been made. (Section II).
4. Decontamination of the primary hot cell has progressed satisfactorily. Levels are now low enough to permit entry by personnel in protective clothing. (Section III).
5. The remaining five fuel specimens (of the eleven mentioned last quarter) have been examined in the SRE hot cell. Diameter changes, estimated total atom burnup and irradiation temperatures are listed. (Section IV).
6. In canning experimental fuels it was found that the amount of NaK added is critical. Experiments have shown that the amount of NaK to be added is between 6.00 and 6.35 cubic inches. (Section IV).
7. The casting of uranium into zirconium thin walled tubes appears promising. Further work with zirconium will be investigated. (Section VI).
8. Slugs were received for the Experimental Fuel Program. They included U-2% Mo, U-1.2% Mo and U-1.8% Mo. (Section X).
9. The conceptual design of the moderator can removal cask is approximately 75 per cent completed. (Section XI).
10. Construction of the 6-inch sodium pump and test loop is complete. The modifications of the 2-inch pump test loop are also complete. (Section XII).

11. Tests of the prototype Mark I control rod revealed that this design is not suitable for use as an SRE regulating rod. The area of difficulty in the Mark I design is the inability of the ball-nut lead screw to operate for more than 10,000 regulating cycles. (Section XIII).
12. During tests on the Mark I safety rod prototype the thimble developed severe leaks. The thimble was modified (Fig. 18) and proved satisfactory. Tests were terminated due to latch failures. (Section XIII).
13. Failure of the freeze seal on the main primary sodium pump resulted in flooding of the pump case. The pump was removed, repaired and reinstalled with an improved gas seal gasket arrangement. (Section XIV).

SECTION A
TECHNOLOGY OF THE SODIUM GRAPHITE REACTOR

The material used in this Section was contributed by the following persons:

W. D. Allen

D. F. Casey

W. P. Corcoran

J. M. Davis

F. L. Fillmore

R. C. Gerber

C. Guderjahn

B. R. Hayward

R. B. Hinze

H. E. Johnson

R. A. Laubenstein

J. E. Mahlmeister

W. K. McCarty

J. J. McClure

W. Morgan

H. N. Royden

G. A. Schneider

J. Stanley

J. Walter

L. Wilkinson

I. FULL-SCALE SGR

A. PLANT CONTROL (W. Morgan)

Steady-state and variable load analysis of the SGR heat transfer system was made. Analyses were made of cases involving combinations of variable load, coolant flow, primary-to-secondary coolant flow ratio, and preheat bypass. All studies are on the basis of constant steam pressure.

Figure 1, 2 and 3 show the conclusions drawn from the study. Figure 1 shows the simple case, where sodium flow rate is varied with load, but the secondary sodium flow rate is maintained equal to primary sodium flow rate, and no bypass controls on the boiler are used. Solutions were obtained using a code developed for the IBM-704 Computer.

If such a program were followed, the consequences for a load change from 10 to 100 per cent are

- 1) T_H (reactor outlet temperature) must change over 150° F .
- 2) T_C (reactor inlet temperature) must change over 80° F .
- 3) ΔT_R (reactor ΔT) cannot be held constant.
- 4) t_{sh} (superheated steam temperature) is fairly constant between 40 and 100 per cent load, but is below normal at loads less than 40 per cent.
- 5) While these results are not necessarily bad in themselves, they indicate that the rate of power change of the plant would be restricted by the allowable rate of temperature change.

Figure 2 shows the result of varying primary-to-secondary sodium flow rate in an effort to hold reactor temperatures constant. The conclusions are:

- 1) Reactor temperatures (T_H and T_C) can be held approximately constant.
- 2) T_4 (secondary loop cold leg temperature) becomes quite low.
- 3) The temperature difference across the cold end of the intermediate heat exchanger ($T_C - T_4$) may be excessive at low loads.
- 4) Superheated steam temperature becomes high at low loads (this can be corrected by attemperation).

Fig. 1. Temperature vs Per Cent Load
($WR/WN = 1.0$, $R = 0$)

Fig. 2. Temperature vs Per Cent Load
(WR/WN = Variable, $R = 0$)

Fig. 3. Temperature vs Per Cent Load
($WR/WN = 1.0$, $R = \text{Variable}$)

Figure 3 shows the results of bypassing part of the feedwater around the preheater (sometimes called economizer) zone of the boiler. Sodium flow ratio is maintained at 1.0. The results are:

- 1) Reactor inlet temperature can be easily controlled at a constant value.
- 2) Secondary sodium cold leg temperature varies only 25° F.
- 3) Reactor outlet temperature is normal at the low loads. The dip of the T_H curve at intermediate loads is attributed to a breakdown of the mathematical model of the preheater zone due to probable early boiling of the feedwater.

B. ALTERNATE CORE CONFIGURATION STUDIES (R. C. Gerber, J. E. Mahlmeister, J. J. McClure, G. A. Schneider, W. D. Allen, W. P. Corcoran and D. F. Casey)

1. Conceptual Plant Layout Studies - 200-Mwe SGR

A conceptual plant layout utilizing vertically mounted and removed intermediate heat exchangers has been completed. The intermediate heat exchanger and primary pump for each loop is combined into one unit which is suspended in a sodium tank. This allows a close coupled reactor and heat transfer system with a plant area for the 200-Mwe SGR about the same as the 75-Mwe plant.

2. Steam Plant Cycle Evaluation

Cost data, expressed in \$/kw vs ΔT of sodium through reactor, has been compiled for steam plants producing 850° F steam and 950° F steam for the following components (see Fig. 4 and 5).

a. Sodium System

- 1) Piping
- 2) Valves
- 3) Pumps
- 4) Surge tanks

b. Heat Exchangers

- 1) Intermediate heat exchangers
- 2) Steam boilers and superheaters

Fig. 4. Steam Plant Costs for 200-Mw Electrical vs Reactor Temperature Differences (850° F Steam)

Fig. 5. Steam Plant Costs for 200-Mw Electrical vs Reactor Temperature Differences (950° F Steam)

c. Steam Plant

- 1) Feedwater heaters
- 2) Piping and pumps
- 3) Steam condensers
- 4) Condenser circulating water pumps
- 5) Water supply and treatment
- 6) Steam plant instrumentation and controls
- 7) Turbo-generators
- 8) Electrical distribution and accessories
- 9) Turbine building.

3. Oxide Fuel for SGR's

Heat transfer and nuclear evaluation of various uranium-oxide fuel element designs is being made. Preliminary results indicate that approximately 210-19 rod oxide fuel elements in a 14-foot diameter core will produce 243 thermal Mw (75-Mw electrical), with a maximum fuel temperature of approximately 4000° F. The required fuel enrichment is 3 per cent. The net fuel cost for 10,000 Mwd/ton exposure and \$30/gm Pu credit is approximately 1.7 mills/e kwhr.

Evaluation of 37-rod cluster and concentric cylinder fuel elements have been started. Studies of the effective resonance integral of U and UO_2 are being made.

II. REACTOR PHYSICS

A. SRE WET CRITICAL ASSEMBLY (C. Guderjahn)

At the start of this experiment the reactor was dry and contained 19 fuel elements. The reactor was heated to 400° F and sodium was added. Fuel elements were then loaded until criticality was reached. The configuration of the core at criticality is shown in Fig. 6. During the loading the counting rate of fission counters placed in channel 2, 4, 65 and 71 was observed. The average of the normalized inverse counting rate of the three symmetrically placed

Fig. 6. Core Configuration at End of Wet Critical Assembly

counters is plotted in Fig. 7 as a function of number of fuel elements loaded. The data from the dry subcritical assembly are given on the same graph for purposes of comparison.

The wet critical mass was determined to be 32.64 ± 0.10 fuel elements at 400°F . The worth of an additional 34th fuel element was found to be $0.38\% \pm 0.03\%$.

B. CONTROL AND SAFETY ROD CALIBRATION (C. Guderjahn)

One of the SRE control rods was calibrated over most of its length from period measurements. A more extensive control rod calibration will be made at a later date over the entire length of a control rod.

The worth of the safety rods was determined by rod drop experiments. One safety rod was found to be worth $1.5\% \pm 0.1\%$ and 4 safety rods together are worth $4.4\% \pm 0.3\%$.

C. REACTIVITY WORTH OF Th-U FUEL ELEMENT (C. Guderjahn)

A Th-U fuel element was substituted for a normal fuel element in three different locations in the SRE. This element contained 5 per cent U^{235} in the form of highly enriched uranium. The reactivity changes as calculated from period measurements were as follows:

<u>*Channel Number</u>	<u>Change in Reactivity (%)</u>
44	-0.062
67	-0.075
73	-0.062
	<hr/>
Average	-0.066

*Refer to Fig. 6, page 19.

During the above measurement, the reactor contained 35 fuel elements and was at a temperature of 385°F . If each of these elements were replaced by a Th-U element, the total decrease in reactivity would be $35 \times 0.066 = 2.3$ per cent. The reactivity decrease calculated for this uranium enrichment was about 2.1 per cent.

Fig. 7. Approach to Criticality of Dry and Wet SRE

D. REACTIVITY STUDIES AND FLUX PERTURBATIONS (F. L. Fillmore)

Two two-group calculations of the SRE critical mass were brought up to date. The results together with experimental values are:

<u>Case</u>	<u>Calculated</u>	<u>Experimental</u>
Dry (20° C)	20.4 fuel clusters	22.2 fuel clusters
Wet (180° C)	28.4 fuel clusters	32.9 fuel clusters
Hot-poisoned (425° C)	32.9 fuel clusters	not available

An evaluation of the reliability of the dry SRE two-group criticality calculations was undertaken. The uncertainty in the calculated critical mass due to assumed uncertainties in various nuclear and physical quantities is shown in Table I. The values given in column two represent an evaluation of the accuracy associated with each quantity. They are based on a combination of the uncertainties of nuclear measurements cited in the literature and an estimate of the probable uncertainty in the calculations due to other effects. Excepting item three, the results cited should not be interpreted as probable errors.

The greatest uncertainty arises from the excess absorption term in the resonance escape probability. The result shown here represents an extreme value of this uncertainty. This result was obtained as the difference in the excess absorption calculated by taking the lethargy spread of resonance neutrons equal to 2.6 and 5.6 respectively, and it seems difficult to justify a change in the lethargy spread of more than this. The value 2.6 for the lethargy spread was used in the critical mass calculations.

It may be reasonable to adopt half of the value shown in Table I for the uncertainty due to the excess absorption term. If it is then assumed that all of the uncertainties listed in Table I are independent, the total uncertainty in the calculated critical mass is obtained by combining the uncertainties quadratically. The result is 23 per cent uncertainty in the critical mass. This value is probably a little high because several of the uncertainties listed in the Table are rough estimates and may represent extreme values rather than probable errors. However, it would appear that on the basis of present information there is an uncertainty of about 20 per cent in the two-group critical mass calculations.

TABLE I

RELIABILITY OF DRY SRE TWO-GROUP CRITICALITY CALCULATIONS

Quantity	Assumed Uncertainty (per cent)	Resulting Uncertainty In Critical Mass (per cent)
1. Age	1.5	1.5
2. Fast effect	10 in $\epsilon - 1$	3.1
3. Eta	1.35	11.3
4. Thermal flux distribution in a lattice cell	5	9.3
5. Absorption cross section		
a) Absolute value (influence L^2 only)	2	1.8
b) Relative value (influences L^2 and f)	0.25	0.4
6. Resonance escape		
a) Effective resonance integral	10	8.3
b) Excess absorption	70	28.8
c) Moderator scattering cross section	2	1.5
7. Transport cross section		
a) Thermal	5.2	4.2
b) Fast	7.3	3.4
8. Axial reflector properties		
a) Absorption cross section	20	1.5
b) Transport cross section	9	2.1

In connection with the uncertainties in the critical mass calculation, further study of the results of the exponential experiments in graphite lattices is being undertaken. The greatest uncertainty is introduced by p , so an attempt is being made to fit the two main parameters involved here, namely the effective resonance integral and $\mathcal{H}_m$, with the exponential experiment results.

Experimental flux measurements are available for these exponential experiments and are used in the calculation of f and L^2 . The methods used to analyze these measurements will be re-examined.

E. DETERMINATION OF SRE POWER AND TEMPERATURE COEFFICIENTS OF REACTIVITY (H. N. Royden)

The SRE moderator temperature coefficient of reactivity appears to be positive on the basis of calculations discussed in a previous quarterly report¹ and on the basis of preliminary measurements. As calculated, the positive moderator coefficient overrides the negative fuel coefficient; therefore, the magnitude of the thermal time constant associated with the moderator temperature becomes important in determining the transient response of the SRE. This time constant has been recalculated with a wide variety of boundary conditions (i. e., heat-transfer coefficients) using slab geometry instead of the actual hollow cylinder. The term "time constant" is used because the transient solution of the heat-diffusion equation consists of a sum of exponential terms, with the coefficient of the first term (which has the longest time constant) being considerably larger than the others. The result is 200 ± 50 seconds, where the variation is caused by the variation in assumed boundary conditions. The time constant is very insensitive to changes in heat-transfer coefficients, but, obviously, not to the assumed graphite thermal conductivity. Radiation damage to the graphite would lengthen this time constant considerably.

The thermal time constant of an SRE fuel rod has also been calculated for a similar large variation in heat-transfer coefficients, and is 2.0 ± 0.5 seconds. The fuel temperature coefficient, which is negative, will dominate the transient behavior of the SRE.

A preliminary measurement of the SRE isothermal temperature coefficient at 400° F has been made. The result is approximately $+2 \times 10^{-5} \Delta k / ^\circ \text{F}$, which is approximately half of the theoretical value at this temperature.¹

F. LONG TERM REACTIVITY CHANGES IN THE SRE (R. A. Laubenstein)

Detailed theoretical calculations on the long term reactivity changes to be expected in the SRE have been initiated. To date, however, only approximate calculations of the reactivity changes have been made. These approximate calculations assume a uniform constant flux distribution and take account of the burnout of U^{235} , the production and burnout of Pu^{239} , the burnout of U^{238} , and the production of fission products poison. This calculation gives the following results:

TABLE II
APPROXIMATE CALCULATIONS OF REACTIVITY CHANGES

E(Mwd/Tonne)	1500	2500	3500
$-\rho_1$ in \$	3.24	4.13	5.43
$-\rho_2$ in \$	1.93	2.86	4.13

$-\rho_1$ = reactivity to override peak Xe^{135} and long term effects

$-\rho_2$ = reactivity to override long term effects only.

A two dimensional diffusion theory calculation of the flux distribution in the SRE has been performed with the NICK-II-Code and the IBM-704 Computer. This calculation allowed for the temperature gradient and consequent variation in k throughout the reactor volume which is expected for full power reactor operation. Further effort on this project will involve the application of these results to the calculation of isotope buildup and burnout and consequent reactivity changes in the SRE.

III. DEVELOPMENT OF HOT CELL FACILITIES AND HANDLING TECHNIQUES

(J. M. Davis)

A. CELL CLEANUP

The primary hot cell was decontaminated to a level which allowed for permissible entry by personnel in protective clothing. This was necessary in order to make equipment changes which became essential due to malfunction and damage caused both by the fire and by inadequate design. The cell service area was also cleaned.

B. EXAMINATION OF NAA-15-4

This MTR-irradiated capsule assembly of experimental fuel materials was disassembled and examined. All required dimensional data was taken except where fuel distortion was too great to permit complete disassembly.

C. PREPARATION FOR EXAMINATION OF NAA-15-5, 15-2 AND HEATERS

Since no equipment changes seemed necessary to examine NAA-15-5 and 15-2, it was not essential to open the cell until these examinations were completed. The interim time was spent in thoroughly decontaminating the service area so that a false ceiling and an expanded change room could be installed. These alterations would materially improve contamination control and, thus, decrease decontamination time, resulting in much more economical operation of the hot cells.

Unfortunately, the examination of the heaters will require a new holding fixture on the milling machine. Therefore, the cell must be cleaned down to a 10-minute entry time, opened, and the fixture installed. For this reason, the heater examination will be scheduled after NAA-15-5 and 15-2. The fixture is now being designed and certain other out-of-cell equipment has been designed and built to improve operations.

D. METALLURGICAL CELL WINDOW

The cell window was reassembled under the direct supervision of the manufacturer's engineer. A major effort was made to gas-seal the interfaces and introduce dry N_2 . The window is now reinstalled in the cell and there does not appear to be any recurrence of the crystalline deposit.

IV. METALLURGY OF SGR FUELS

(B. R. Hayward, J. Walter and L. Wilkinson)

A. FUEL IRRADIATION TESTS

Irradiations of fuel specimens have been conducted in the MTR under thermal stress and temperature conditions approximating the SRE conditions. The results on the first six of the eleven dilute uranium alloy and standard unalloyed SRE fuel samples irradiated were reported in the previous report.² The remaining five specimens have been examined in the SRE hot cell. Of these five, only two could be removed from the capsule for measurement. These two were removed, measured, and photographed.

Photographs of an unirradiated specimen and the two irradiated specimens are shown in Fig. 8-A, 8-B and 8-C. A growth at the end of the specimens and a fairly smooth surface is evident on both of the irradiated specimens. This appearance is typical of the specimens previously examined. The reasons for the growth at the top end of the specimen is not understood.

The diameter change on the smooth portion of the specimens, the estimated total atom burnup, and the irradiation temperature are listed in Table III. The results on these specimens, which were irradiated to less burnup at a higher temperature than the previous specimens, correlate with the results previously reported for U-1.2 w/o Mo powder-compacted fuel.

TABLE III

EFFECTS OF IRRADIATION (POWDER-COMPACTED U-1.2 w/o Mo SPECIMENS)

Figure	Specimen* Composition	Estimated Burnup	Max. Irradiation† Temp.	Max. Diameter Increase
8-B	U-1.2 w/o Mo	0.2	1250° F	8 per cent
8-C	U-1.2 w/o Mo	0.2	1100° F	11 per cent

*The original specimens were nominally 0.375 in. diameter by 1.500 in. long.

†Temperatures were measured along the axis of the specimens 0.500 in. above the bottom end of the specimen.

Fig. 8-A. Typical Specimen Before Irradiation

Fig. 8-B. Effect of Irradiation on a Powder-Compacted 1.2 w/o Mo-U Alloy Specimen. Estimated Total Atom Burnup 0.2%. Maximum Measured Irradiation Temperature 1250° F.

Fig. 8-C. Effect of Irradiation on a Powder-Compacted 1.2 w/o Mo-U Alloy Specimen. Estimated Total Atom Burnup 0.2%. Maximum Measured Irradiation Temperature 1110° F.

B. CANNING OF EXPERIMENTAL FUELS

The SRE fuel rod consists of a 0.010-in. wall stainless-steel jacket filled with fuel slugs bonded to the jacket with 0.010 in. of NaK. There is space for excess NaK and gas above the fuel slugs. If the SRE slugs grow in volume as indicated by the MTR fuel irradiation tests, the burnup in the SRE fuel rods will be limited to the burnup required for the center fuel slug to fill the 0.010-in. NaK annulus. The effect of the increase in the fuel rod internal pressure due to the filling of the NaK annulus has been analyzed.

The results indicate the amount of NaK added is critical. More than 6.35 cu. in. of NaK will result in excessive stresses in the jacket to end cap welds due to too little space for NaK expansion. Less than 6.00 cu. in. of NaK would not provide sufficient excess NaK to bond the slugs if the rod leaks NaK. A revision of the loading procedure is being devised to insure that the amount of NaK in the experimental fuel rods fall within these limits.

The dilute uranium alloy fuel slugs for the experimental mixed cluster have been cleaned and outgassed for canning. The slug surface preparation was varied to determine the effect of the slug surface condition on fuel rod performance. Pickling or cleaning with acetone was substituted for the standard electropolish on selected fuel rods. Final canning of these rods is awaiting the revision in the NaK loading procedure.

C. THERMOCOUPLED FUEL ELEMENTS

Thermocoupled fuel elements will be used to determine maximum SRE operating conditions when limited by fuel temperature and to provide a temperature map of the core. These elements each contain two rods with clad thermocouples extending axially into the fuel slugs to different depths. Six thermocoupled fuel elements made up of standard unalloyed SRE fuel have been assembled. Five of these elements have been inserted into the SRE core and read $\pm 2^\circ \text{F}$ of the known sodium temperature at 400°F . All of the experimental fuel elements will contain thermocouples.

V. METALLURGY OF BREEDER FUELS

(J. Stanley, B. R. Hayward and J. Walter)

Thorium-uranium alloys are being studied as promising high temperature and long burnup breeder fuels.

To check the uniformity of enrichment from slug to slug in the extruded Th 5.4 w/o enriched U SRE fuel slugs made for irradiation in the SRE, danger coefficients were made in the water boiler neutron source. The indicated variations in enrichment were less than ± 0.05 w/o U^{235} which is less than that indicated by chemical analysis.

Four Th-5.4 w/o U fuel elements will be used in the SRE for nuclear measurements and long time irradiations. Rods for three of these elements have been canned and one element completely assembled. The canning of the rods for the fourth element is awaiting completion of a modification in the NaK bond loading apparatus.

The slugs for eight Th-5.4 w/o U rods to be included in the mixed experimental fuel clusters have been prepared for canning. Variables in the slug preparation will be studied by irradiating rods containing as received slugs; outgassed, as received, slugs; and pickled and outgassed slugs.

VI. ADVANCED FUEL MATERIALS

(J. Stanley)

Effort has been directed toward the development of several methods for producing 3/8-in. diameter lengths of Th-10 w/o U slugs. There has been some success in casting a homogeneous 1-3/4-in. diameter by 6-in. long Th-10 w/o U sound ingot (~ 2 kg). However, this is considered too large and too involved a process for economically making 3/8-in. diameter by 1-1/2-in. long Th-10 w/o fully enriched uranium specimens. A rapid pour type crucible, distributor, and 1/2-in. diameter mold arrangement have been designed and used. A "gang" casting of four 1/2-in. diameter by 4-1/2-in. long slugs was made. Two rods

have been sectioned for chemical and metallurgical analyses and the other two rods have been machined to 0.450-in. diameter for subsequent swaging studies of metals behavior during hot or cold fabrication operations.

After an investigation of pour temperatures, melting cycles, charge preparation, and type wash for graphite crucible, it was found that a pour temperature of at least 1750° C, a 30-minute melting cycle regardless of charge preparation, a graphite crucible washed with BeO, and a heated mold is required to repeat satisfactory casting performance.

In addition several investigations have been made in casting molten uranium into zirconium and into stainless-steel calorized thin walled tubes. Zirconium results appear encouraging and further work will be explored. Another installation associated with this work had experimented similarly and found that zirconium cans behaved best in regard to the bonding of a uranium core with a thin wall tube.

The centrifugal casting unit is complete and has been received. Exploratory work will be performed on the casting of normal uranium slugs prior to any work on Th-U alloys.

VII. CORROSION IN SODIUM

(R. B. Hinze and H. E. Johnson)

Exposure of metal specimens in the materials evaluation facility at the SRE began in April. Zirconium, Type 304 stainless steel, and beryllium samples are being exposed in the primary (radioactive) sodium system to determine their corrosion characteristics.

Examination of zirconium specimens will permit evaluation of the behavior of the moderator cans in the core. Oxygen concentration in the primary sodium system will be monitored by determining the presence of oxide film on polished zirconium samples. This will be the basis for determining the performance of the hot trap as a gettering device for oxygen.

In the core of the SRE, beryllium is exposed to sodium in the neutron source element and in two corner channel elements used for temperature measurements.

Therefore, beryllium samples have been included in the initial loading of the hot thimble. Type 304 stainless steel specimens have been included to determine changes in the major structural material of the SRE.

The materials exposure facility is located in the main gallery of the SRE. The flow diagram (Fig. 9), shows the relation of this facility with the main, intermediate heat exchanger and the reactor core. Sodium at reactor outlet temperature is tapped off the pump discharge and flows up the hot thimble. Cool sodium at core inlet temperature is fed to the cold thimble. Figure 10 shows the spindle with initial loading for the hot leg. A full spindle has been placed in each 2-1/2 in. exposure pipe. Holders, mounting four specimens each, will be added or removed from the thimbles using the fuel handling coffin. The insert in Fig. 10 shows the variety of samples: beryllium sheet in perforated containers, stainless steel, and four kinds of zirconium specimens (vapor blasted, lapped for fatigue testing, welded, and formed moderator can side panel).

VIII. ENGINEERING EVALUATION OF GRAPHITE

(W. K. McCarty)

Facilities for evaluating the extent and nature of radiation-induced degassing of SRE graphite were placed in the reactor core. These consisted of two moderator elements and two corner channel dummy elements with cans vented through the reactor top shield. Extreme care was taken in testing the facilities for gas tightness before installing them in the core. Communication between the reactor plenum volume and the reactor room was eliminated by development of adequate static seals. Gas volumes were calibrated and each facility was filled with a prescribed quantity of helium.

The volume of gas evolved from the graphite, during reactor core heat-up, was in agreement with previously measured data on thermal degassing. Reversible pressure changes beyond this were in close parallel with the perfect gas expansion expected at the temperature measured by thermocouples in each facility. No unexpected effects upon gas pressure over the graphite were observed during subsequent filling of the core volume with sodium.

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Fig. 9. Flow Diagram of Materials Exposure Facilities

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Fig. 10. Sample Loading SRE Materials Exposure Facility

SECTION B
SODIUM REACTOR EXPERIMENT

The material used in this Section was contributed by the following persons:

R. W. Atz

H. R. Boise

W. A. Davis

J. J. Droher

E. O. Dryer

S. Elchyshyn

R. B. Hall

D. H. Johnson

A. E. Miller

E. C. Phillips

J. F. Stolz

IX. LAND, UTILITIES AND BUILDINGS

(J. F. Stolz)

Gallery shield block anchorages were modified to forged anchorages for the purpose of providing a more appropriate factor of safety. In connection with this effort a hoisting rig was designed consisting of an equalizer beam and appropriate slings and spreaders.

To provide easier access to the operating machinery on top of the SRE fuel handling coffin, a caged ladder and catwalk was designed and installed.

A filter storage house was designed and installed on the roof of the SRE reactor building for storing fresh filters for the exhaust fans of the high bay roof.

X. FUEL ELEMENTS

(J. J. Droher and S. Elchyshyn)

Additional fuel elements were assembled at the Engineering Test Building and transferred to the SRE. The following numbers of fuel elements have been transferred to the SRE: 41 Standard Fuel Elements, 6 Thermocoupled Fuel Elements and 1 Th-5.4 w/o U Fuel Element. Each thermocoupled fuel element contains five standard fuel rods and two rods with thermocouples sealed in the upper closure and extending through axial holes drilled to various depths in the fuel slugs. These thermocouples will enable measurements to be made on the temperature profile through the reactor core. The Th-U element was used during dry sub-critical measurements.

One of the standard fuel elements was damaged during fuel handling operations due to equipment failure; it was returned to the Engineering Test Building. All but two of the 84 slugs can be reloaded.

The fuel element lower guide and spacers were modified to eliminate the possibility of hang-up between the fuel element and the coiled wire guide recently added to the moderator cans.

Sixteen batches of various alloyed uranium slugs and two batches of cast unalloyed uranium slugs were received at the Engineering Test Building. The Alloyed slugs include eight batches of U-2 w/o Zr, four batches of U-1.2 w/o Mo, and four batches of 1.8 w/o Mo. These slugs will be used in fuel rods for the Experimental Fuel Program. Most of the slugs were electropolished but selected batches were cleaned in acetone or pickled. These slugs have been outgassed and are ready to be loaded into fuel rods.

XI. MODERATOR CAN FABRICATION AND TESTING

(E. O. Dryer)

The conceptual design of the moderator can removal cask and removal techniques as illustrated in Fig. 11 is about 75 per cent completed. Alternative designs which involve shielding the operator instead of the moderator can are under study.

Before the final equipment is available, it may be necessary to provide an interim method of removing cans at lower levels of radioactivity. This study is now underway.

XII. HEAT TRANSFER

(R. W. Atz)

A. SIX-INCH PUMP TEST LOOP

The construction of the 6-inch sodium pump and the test loop has been completed. The pump assembly is shown in Fig. 12 and the loop before insulation is shown in Fig. 13. A part of the construction work included the installation of a heating transformer (Fig. 14). This transformer will be used to assist in preheating the loop for filling with sodium and it can be used for maintaining elevated sodium temperatures in the system during loop operation. The use of this transformer has eliminated the need of 90 per cent of the resistance heaters that were formerly used for preheating the loop. This method of heating the sodium in the

Fig. 11. Moderator Can Removal Cask (Conceptual Design)

9693-54529B

Fig. 12. Six-Inch Pump Assembly

9693-54525D

Fig. 13. Six-Inch Pump Loop

9693-54528A

Fig. 14. Heating Transformer (for the Six-Inch Pump Test Loop)

loop permits the application of heat without the necessity of having the sodium circulating in the system. The immersion heaters require that the sodium be kept flowing to dissipate the heat. It will then be possible to measure the cooling loads on the sodium freeze seals when the pump is not operating.

The pump is being altered to permit preliminary operation using water as a pumped fluid. It is desired to measure accurately the pressure retained by the shaft freeze seal of the pump under various pumping conditions. This apparatus will also be used to perform a complete hydraulic test upon the pump.

Water will also be used during the preliminary phase of the pump modification program. There are parts of the mechanical support of the pump that should be changed. These revisions can be done most conveniently by using low carbon steel and testing with water.

B. TWO-INCH PUMP TEST LOOP

The modification of the 2-inch pump test loop for use on the alternate materials of construction program was completed. The oxygen concentration is being measured with a plugging meter and a diffusion cold trap is being used to reduce the amount of oxide present in the system. A test section of pipe containing specimens for the alternate materials of construction for sodium service has also been installed. The objective of the test is to study the effect of sodium upon materials which may replace Type 304 stainless steel.

Two heating transformers have been installed to assist the resistance heaters in preheating the system preparatory to filling with sodium.

The experiment was started with 87 ppm oxide present. After 250 hours of operation, a weld failure in the test section forced termination of the test. The last 80 hours of the test were conducted at temperatures over 1000° F with an oxide concentration of 34 ppm.

XIII. INSTRUMENTATION AND CONTROL

A. PRODUCTION DRIVES (A. E. Miller)

One single-speed drive and 2 two-speed drives required for the SRE remain to be completed. One of each type of drive is a reactor spare. The prototype

two-speed drive is in temporary use at the SRE, thus permitting satisfactory reactor operation.

B. MARK I CONTROL ROD LEAD SCREW DEVELOPMENT (S. Elchyshyn, H. R. Boise and A. E. Miller)

Life testing of the prototype Mark I control rod revealed that this design is not suitable for use as an SRE regulating rod.¹ The inability of the ball-nut lead screw to operate for more than 10,000 regulating cycles restricts them to shim rod use in the SRE. Tests of the Mark II prototype demonstrated that this design is adequate for regulating rod use,³ however, it is inconvenient because the heavy support housing located above the reactor shield must be removed before fuel can be handled. The lead screw development program is therefore being continued.

Screws of various materials and types will be tested including a rolled and ground Haynes 25 ball screw. In addition, the possibility of continuously lubricating the screw with dry lubricants will be investigated.

During this period a screw test apparatus was assembled and a number of preliminary tests designed to demonstrate the suitability of the apparatus were performed.

1. Test Facility

The test facility consists of a furnace pipe, a 70-lb weight, a reversible drive mechanism and a worm screw limit switch device for limiting the travel of the test screw. The furnace pipe, a 4-inch pipe 13 feet long made up of three flanged sections, is heated by resistance heaters in the test length (Fig. 15). The screw to be tested is suspended in the furnace pipe from the drive shaft which in turn is supported by a thrust bearing. The 70 lb weight is attached to the support nut through a universal joint. The pins of this flexible joint which operates at the test temperature (1100° F) are cold worked Haynes Alloy No. 25 and the mating holes are Haynes Stellite No. 6. A three phase 1/4 hp 180-rpm double ended gear reducer motor drives the drive shaft, and through a miter gear reduction drives a 1-inch diameter 6-inch long screw. This screw drives a guided nut through the 6-inch stroke as the test nut traverses its 14-inch test length. The guided nut in conjunction with adjustable limit switches serves to reverse the motor at the ends of the stroke.

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Fig. 15. Furnace Pipe

The 70-lb weight, equal in weight to the control rod poison column assembly is keyed to the furnace pipe. A rectangular steel bar welded to the bottom of the weight passes through a gimbals mounted brass plate that contains a mating rectangular hole. Thus the weight and the attached test nut is made to travel vertically without rotating. Figure 16 shows an assembly prior to insertion into the furnace pipe.

Rotating O-ring seals maintain the NaK bubbled helium atmosphere in the furnace pipe.

2. Tests

The first tests performed were designed to establish a method of providing continuous dry lubrication for ball-nut screws. To accomplish this a cylindrical can containing molybdenum disulfide was mounted above the test nut (Fig. 17). Graphite wiper nuts at the top and bottom of the lubricant container prevents excessive loss of lubricant.

Because a rolled ball-nut screw specimen was not available a section of a previously used, ground Haynes No. 25 screw and a graphite nut were used in the preliminary tests. The first few tests were unsuccessful because of locking action between the support nut and the wiper nuts and because of insufficient clearance between the graphite nuts and Haynes No. 25 screw.

The first successful run was made at room temperature without an upper wiper nut and with the lower wiper nut and support nut replaced by one graphite nut. The next test was conducted at elevated temperatures. The screw was operated for 16 hours at 830° F and for 4 hours at 1100° F. Neither the nut nor the screw exhibited any damage.

The third and final test conducted during this quarter was run without incident for a total of 500 hours at 1100° F. Twice during this run the temperature was reduced to room temperature and again raised to the test temperature. Examination of the nut and screw showed them to be in excellent condition.

In none of the three tests was there any evidence of jamming because of buildup of lubricant between the nut and screw. While these tests were designed to develop a method of continuously lubricating a ball nut screw, the results thus far indicate that a lubricated graphite nut and Haynes No. 25 screw

Fig. 16. Test Assembly (Mark I Control Rod Lead Screw)

9693-55284B

Fig. 17. Detailed View of Screw Test Apparatus (Upper End)

might be satisfactory for the Mark I control rod lead screw. Further tests designed to substantiate this and to develop a non-locking upper wiper are in progress.

C. MARK I SAFETY ROD PROTOTYPE (E. C. Phillips and W. A. Davis)

The Mark I safety rod prototype was operated through 946 cycles at 800° F. The test was interrupted twice. The first interruption occurred when the drive shaft worked loose from the safety rod drive motor. The second interruption was caused by the failure of a cannon plug. The test was terminated by the failure of a second cannon plug.

During this test, the thimble safety rod developed a severe leak. The leak was caused by cracks in the thimble at the juncture of the thin walled section to the heavy walled section. Cracks occurred in the heat affected zone immediately adjacent to the weld bead.

The thimble was modified so that the thin walled section just below the weld to the heavy walled section is supported by a heavy walled sleeve (see Fig. 18).

The Mark I prototype with the modified thimble was cycled 300 times under a simulated reactor gradient with a peak temperature of 1000° to 1100° F. The test was terminated when the latch failed to lock. Graphite fragments from a broken shield ring prevented the latch from working.

The Mark I prototype with the modified thimble was cycled an additional 397 times under a simulated reactor gradient with a peak temperature of 1000° to 1100° F. The four piece graphite shield section was replaced by a single piece of graphite. The test was terminated when the latch failed to lock. The latch was damaged by two fragments of boron nickel from a poison ring.

The Mark I safety rod prototype with the modified thimble was reassembled for a third test with two of the boron nickel poison rings replaced. The prototype was cycled 521 times under a simulated reactor gradient with a maximum temperature of 1000° to 1100° F. The test was terminated by the failure of the ball nut extension latch.

Two additional tests were made consisting of 120 and 166 cycles each. Both tests were terminated by failure of the ball nut extension latch. In these two tests, an aluminum oxide flame plated latch and chromized drill rod keys were used.

Fig. 18. Modified Thimble for the Mark I Safety Rod Prototype

D. SRE MARK II PROTOTYPE SAFETY ROD (E. C. Phillips and W. A. Davis)

During preliminary hot tests of the Mark II prototype safety rod a pin in the pull tube sheared disconnecting the poison column from the piston assembly. After the failure of the pin, all sections of the pull tube were welded together to strengthen the pull tube.

It was necessary to replace a drive motor on the Mark II safety rod prototype after 193 drops at a temperature of 1000° to 1100° F.

A total of 800 drops at temperatures of 1000° to 1100° F have been made with the Mark II safety rod prototype. The pull tube was found to be galled over its entire length where it had rubbed in the open in the lower end of the shield section, but the rod was still in operating condition.

E. STRAIN TESTS ON SRE MARK II SAFETY ROD (E. C. Phillips and W. A. Davis)

The Mark II safety rod prototype is being modified for strain tests.

F. SRE MARK II PRODUCTION SAFETY RODS (E. C. Phillips and W. A. Davis)

Five Mark II production safety rods were assembled and tested for time response. Time response was satisfactory and met required drop times. Four Mark II safety rods were installed in the reactor.

XIV. REACTOR SERVICES

A. PREOPERATIONAL TESTING (D. H. Johnson)

The core and primary systems were preheated, filled with sodium and checked for tightness and proper operation of components. No leakage of sodium had occurred. Sodium was circulated through the primary and secondary systems at ~ 350° F.

Criticality was achieved at 12:19 PM April 25, 1957 with a loading of 33 fuel elements.

B. REACTOR OPERATIONS (D. H. Johnson)

Fuel elements, dummy elements and core heaters have been successfully washed in the fuel cleaning cells; the resulting residue was successfully transferred to the liquid waste storage.

Control rod calibration was started, but was stopped when the main primary sodium pump freeze seal failed, flooding the pump case. The helium pressure in the pump case had been reduced to conserve helium leakage from the top flange in an attempt to complete the control rod calibration before removing the sodium pump.

All the sodium was drained from the reactor. The main primary sodium pump was removed, repaired and reinstalled. An improved gas seal gasket arrangement was installed and no helium leaks were detected.

The reactor primary sodium loops were refilled with sodium and the reactor prepared for continuation of nuclear tests. Control rod calibrations were performed with the reactor critical.

Tetralin was detected in the auxiliary secondary pump, and syntron oil was found in the main secondary pump case. Both were removed, cleaned and reinstalled. The secondary system loops were refilled with sodium and preparations made for cold trapping.

C. ELECTRICAL DESIGN (R. B. Hall)

A high percentage of "burn-out" of the flexible resistance heaters on the SRE pump casings necessitated replacement with tubular resistance heaters. Failure of the flexible heaters is contributed to the poor quality of the heaters. The lead connection on the heater terminals did not meet the specified temperature requirements.

An electrical automatic starting feature was installed on the stand-by gasoline engine for the tetralin pumps. During an electrical fault or failure of the SRE emergency power system, the stand-by gasoline engine will automatically start up and continue the tetralin pumping action.

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